NON-PUBLIC?: N
ACCESSION #: 8907210308
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 1 PAGE: 1 of 4

DOCKET NUMBER: 05000272

TITLE: Rx Trip #13 Steam Gen. Lo-Lo Level Due To An Equipment Design Concern
EVENT DATE: 06/19/89 LER #: 89-027-00 REPORT DATE: 07/17/89

OPERATING MODE: 1 POWER LEVEL: 045

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. J. Pollack, LER Coordinator TELEPHONE: 609 339-4022

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: no

ABSTRACT:

On 6/19/89 at 2100 hours, a Reactor Trip on No. 13 Steam Generator (S/G) "Low-Low Level" occurred. The No. 13 main steamline isolation valve, 13MS167, had closed. Prior to the event, reactor power was being increased 3% per hour. At the time of the event, a post maintenance surveillance for the 12MS18 Main Steamline Bypass Stop Valve was in progress. The root cause of this event has been attributed to inadequate design of the continuity check circuitry for the MS167 valves. Surveillance testing of valve 12MS18 was in progress prior to the event. When the Solid State Protection System (SSPS) Train "A" output interface cabinet switch was turned to "operate output", the 13MS167 valve closed followed by the trip. The "operate output" function causes closure of the MS18 valve while checking continuity of the MS167 valve closure circuit. This design can cause inadvertent closure of the MS167 valves as occurred during this event and a similar Unit 2 event on 4/11/89 (ref. LER 311/89-008-00). A design change has been implemented which corrects the circuit design concern by adding a contact which prevents the 74-3A relay from resetting during the testing of the MS18 valves. This contact does not prevent the MS167 valves from functioning in the event of a valid main steam isolation signal. A similar Unit 2 design change will be

implemented during the next outage of sufficient duration. In the interim, a new procedure will be issued which addresses the surveillance testing of the MS18 valves. The Unit 1 74-3A relay was replaced and will be sent to the vendor for evaluation.

END OF ABSTRACT

TEXT PAGE 2 OF 5

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as xx!

IDENTIFICATION OF OCCURRENCE:

Reactor Trip On No. 13 Steam Generator Low-Low Level Due To An Equipment Design Concern

Event Date: 6/19/89

Report Date: 7/17/89

This report was initiated by Incident Report No. 89-386.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 45% - Unit Load 388 MWe

DESCRIPTION OF OCCURRENCE:

On June 19, 1989 at 2100 hours, a Reactor Trip on No. 13 Steam Generator (SIG) "Low-Low Level" occurred. The No. 13 main steamline isolation valve, 13MS167, SB! had closed. Prior to the event, reactor power was being increased at a rate of 3% per hour.

At the time of the event, a post maintenance operability retest (procedure SP(0)4.0.5-V) for the 12MS18 Main Steamline Bypass Stop Valve was in progress. The valve diaphragm had recently been replaced.

The Unit was stabilized in Mode 3 (Hot Standby) and at 2150 hours, on June 19, the Nuclear Regulatory Commission was notified of the actuation of the Reactor Protection System JC in accordance with Code of Federal Regulations 10CFR 50.72(b)(2)(ii).

APPARENT CAUSE OF OCCURRENCE:

The root cause of this event has been attributed to inadequate design of the continuity check circuitry for the MS167 valves.

As stated above, surveillance testing of valve 12MS18 was in progress prior to the event. When the Solid State Protection System (SSPS) JG! Train "A" output interface cabinet switch was turned to "operate output", the 13MS167 valve closed followed by the trip. The "operate output" function causes closure of the MS18 valve while checking continuity of the closure circuit for all four MS167 valves. This design can cause inadvertent closure of the MS167 valves as occurred during this event and a similar Unit 2 event on April 11, 1989

APPARENT CAUSE OF OCCURRENCE: (cont'd)

(reference LER 311/89-008-00).

A review of this event and Similar prior events was conducted by system Engineering. As Previously indicated, the root cause of the MS167 valve closure is due to the inherent design of the continuity checking circuitry.

During the Unit 1 troubleshooting investigation, the test circuit of which the 74-3A relay (same relay as the Unit 2 74-4A relay) is part of, operated satisfactorily in 32 out of 33 tests. In the one test it did not operate satisfactorily, the voltage at the reset coil of the 74-3A relay remained high enough for a sufficient amount of time to allow the relay to go to the reset position.

The design of the continuity check circuitry includes the use of a "light bulb" in series with the 74-3A relay (an operate reset latching relay) and a time delay relay. If the 74-3A relay resets it will result in main steam isolation. The circuit light bulb is intended to reduce the voltage sufficiently to prevent the triggering of the 74-3A and time delay relays. The original assumption was that this voltage reduction occurs instantaneously when the switch contact is closed. Upon further investigation it has been determined that the 74-3A and time delay relay coils prevent instantaneous current flow, subsequently preventing line voltage reduction. The 74-3A relay will therefore see 125 VDC across it until current does flow. If the voltage at the reset coil remains at this high level for a sufficient amount of time, the relay will go to the reset position and main steam isolation will occur.

The Unit 2 event root cause was attributed to the 74-4A relay (same as the Unit 1 74-3A relay). It was found to be defective.

ANALYSIS OF OCCURRENCE:

The main steam isolation function is designed to minimize the positive reactivity effects of the Reactor Coolant System AB cooldown associated with the blowdown from a steam line rupture and to limit the pressure rise within containment in the event the steam line rupture occurs within containment.

The MS18 valves are part of the main steam isolation function as per the Updated Final Safety Analysis Report. The Technical Specifications specifies that surveillance testing of the main steam isolation actuation circuitry be performed every 62 days on a staggered basis. In addition, after completion of maintenance activities, post maintenance operability testing is performed to ensure operability. As stated in the Description of occurrence section, the 12MS18 valve operability testing was in progress when closure of the 13MS167 valve occurred. This valve closure lead to a low-low level condition in No. 13 S/G which generated the trip

ANALYSIS OF OCCURRENCE: (cont'd)

signal.

The Low-Low S/G level reactor trip prevents operation with the S/G water level below the minimum volume required for adequate heat removal; thereby preventing the loss of the reactor heat sink. The trip is actuated on two out of three low-low water level signals in any S/G. The setpoint ensures that there is adequate inventory in the S/Gs, at the time of the reactor trip, to allow for any possible starting delays of the Auxiliary Feedwater (AFW) System BA ; thus preventing S/G dry-out and the Reactor Coolant System (RCS) (AB)
hermal and hydraulic
transients that would be associated with a loss of the heat sink following quick closure of the MS167 valve.

The 11-14AF21 valves (S/G Aux Feedwater Level Control Valves) functioned as designed during this event; however, after the reactor trip, it was observed by Operations personnel that the 14AF21 valve was apparently leaking by after receipt of a closure signal. Subsequently, the 14AF20 valve (#14 SIG Aux Feedwater Control Isolation Valve) was manually closed. It was used as the S/G level control valve until the return to service of the 14AF21 valve. Investigation revealed that the airline to the valve actuator was broken. The airline was subsequently replaced and the valve functioned according to design. During power operation, the 11-14AF21 valves are closed with a 95% valve demand (open) signal applied in order to provide pump runout protection. This signal is based on the discharge pressure of the Motor Driven Auxiliary Feedwater Pumps in comparison with the S/G secondary side pressure. Therefore, Auxiliary Feedwater flow to the S/Gs will occur in the event of a reactor trip.

The Reactor Protection System (RPS) JC functioned as designed, and the heat sink was maintained. The RCS has been designed to withstand the thermal and hydraulic effects of four hundred (400) trips from full power. This trip was well within the design limits of the system. This occurrence involved no undue risk to the health or safety of the public; however, due to the automatic actuation of the RPS, this event is reportable in accordance with Code of Federal Regulations 10CFR 50.73(a)(2)(iv).

CORRECTIVE ACTION:

Design change 1EC-3008 has been implemented. It corrects the circuit design concern by adding a contact which prevents the 74-3A relay from resetting during the testing of the MS18 valves. This contact does not prevent the MS167 valves from functioning in the event of a valid main steam isolation signal.

A similar Unit 2 design change will be implemented during the next outage of sufficient duration. In the interim, a new procedure will be issued which addresses the surveillance testing of the MS18 valves. This procedure replaces the surveillance testing originally addressed in the procedure for slave relay testing. The procedure will address measures designed to prevent main steam isolation as a result of the surveillance.

CORRECTIVE ACTION: (cont'd)

The Unit 1 74-3A relay was replaced and will be sent to the vendor for evaluation.

LK Miller/
General Manager -
Salem Operations

mjp:pc
SORC Mtg. 89-074

ATTACHMENT 1 TO 8907210308 PAGE 1 OF 1

PSEC&G

Public Service Electric and Gas Company P.O. Box E Hancocks Bridge, New Jersey 08038

Salem Generating Station

July 17, 1989

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-70
DOCKET NO. 50-272
UNIT NO. 1
LICENSEE EVENT REPORT 89-027-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73(a)(2)(iv). This report is required within thirty (30) days of discovery.

Sincerely yours,

L. K. Miller
General Manager -
Salem Operations

mjp:pc

Distribution

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